Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/92
FRANCE
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

A NEW ORDERED STRUCTURE CREATED BY NEUTRON IRRADIATION

by M.M. L. NEEL
D. DAUTREPPE
J. PAULEVE
J. LAUGIER
R. PAUTHENET
A. MARCHAND

(Commissariat à l'Energie Atomique, France)

Neutron irradiation at 300°C in the swimming pool reactor MELUSINE of an Fe-Ni (50 p. 100 at) alloy yields by the acceleration of the diffusion process, a new ordered phase of this alloy, the same type as AuCu.

Evolution is traced by magnetic anisotropy, electrical resistance and X-ray measurements. The surstructure rays obtained confirm the quadratic structure of the AuCu type. The critical temperature determined by the measurement of the electrical resistance in the reactor is about 320°C $\pm$ 2°C. Studies of monocrystals irradiated in a magnetic field have given precise details about the magnetic behaviour of this new alloy which is arranged in small crystallites whose general configuration favorises its orientation in the direction of the magnetic field applied during irradiation.

The interest of this study consists in the manifold industrial applications of iron and nickel alloys.

Translation provided by courtesy of the submitting Government

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/93
FRANCE
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

HEAT TRANSFER WITH ORGANIC FLUIDS *

by F. LANZA - EURATOM ISPRA
LESAGE - EURATOM ISPRA
J. VILLENEUVE - Service des Transferts Thermiques, C.E.N. Grenoble
R. RICQUE - Service des Transferts Thermiques, C.E.N Grenoble
M. COURTAUD - Service des Transferts Thermiques, C.E.N Grenoble
F. MARTIN-LEFEVRE - Service des Transferts Thermiques, C.E.N. Grenoble

(Commissariat à l'Energie Atomique France)

Using a test channel with a circular cross-section of 12 mm internal diameter the following heat transfer correlation, valid to about $\pm$ 6 p. 100, was determined for the terphenyls OMP and OM2 with 0, 10, 20 or 30 p. 100 of high polymers : $Nu_b = 0.00835\ Re_b^{0.9}\ Pr_L^{0.4}$. This correlation was established with fluxes ranging from 50 to 100 W/cm^2, temperatures from 290 to 420°C and Reynolds number from 30 000 to 350 000.

8 long-term tests were carried out (250 to 1000 hours continuous running) to determine how deposition on the heating surfaces affects the exchange coefficient. The loops is made of stainless steel except for one part in soft steel. No deposits are observed even if the terphenyl contains 30 ppm of chlorure, while on the other hand deposition is appreciable when heavy substances containing combined oxygen are added.

* with the help of MM. A. MICHEL and A. TEYTU in building the loops

Translation provided by courtesy of the submitting Government

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/94
FRANCE
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

CO_2 CIRCUITS IN NUCLEAR POWER STATIONS

by

Mr. Edmond ROBERT — ELECTRICITE de FRANCE : General Nuclear Studies Division, Head of "Equipment Techniques" Department.

Mr. Jean HUSTACHE — ELECTRICITE de FRANCE : No 1 Nuclear Region, Head of Engineering Department.

Mr. Philippe SEVIN — ELECTRICITE de FRANCE : No 2 Nuclear Region, Head of Engineering Department.

This report deals with the typical equipment employed in the primary CO_2 circuits of French nuclear power stations using carbon dioxide gas as coolant. This equipment consists essentially of :

- Heat exchangers
- Fans
- Units peculiar to the circuit (obturators, compensating devices).

The heat exchangers associated to the reactor of a nuclear power station of the natural uranium-graphite CO_2 sequence consist in all plants now built or now being completed (Chinon, EDF 1, EDF 2 and EDF 3) of a certain number of identical units working in parallel.

A brief description of the different types is given, together with a list of advantages of this type of solution.

Translation provided by courtesy of the submitting Government

In new power stations of the integrated type (EDF 4) the heat exchanger is housed within the pressure-vessel and thus forms a discrete monobloc unit. Because of this, it has different structural and operational characteristics, which are also mentioned.

As regards the blowers, first and foremost, the vital part played by these in a nuclear power station is explained. Indeed, these machines must operate both reliably and efficiently.

The way in which the best-suited type of machine is chosen, how it is to be driven, the possibilities of adapting it to suit the characteristics of the circuit and the principal problems arising in installing the machines are then examined.

The various solutions adopted by Electricité de France for the different power stations are reviewed, and a list is given of all the difficulties encountered during development and starting up.

Lastly, special mention is given of the various characteristics of the blowers employed in integrated solutions.

Bellows and obturators are also mentioned. These primary circuit devices are difficult to design and implement.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/95
FRANCE
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

TECHNIQUE FOR VORTEX TYPE FLOW WITH PHASE SHIFT IN WATER REACTORS

by C. FOURE D. EIDELMAN C. MOUSSEZ

. S.N.E.C.M.A.

This report is based on work which has been carried out for the most part under a contract with the EURATOM-U.S.A. Joint Committee.

In the first place it has been shown that the longitudinal vortices which are contiguous and coherent can be produced and preserved in channels having different cross-sections, in particular those occuring in boiling water nuclear reactors where the fuel elements are placed at the nodes of a square lattice. Each elementary rootex is produced by a twisted metallic strip placed in the channel involved.

The pressure drop and the proportion of empty space during two-phased flow have been determined as a function of the main parameters which can apparently by regrouped in the form proposed by Martinelli and his co-workers.

By means of these twisted strips it has been possible to increase the heat applied to a boiling water current by 25 to 40 per cent for the same pumping power, depending on whether on operates at 70 bars or at atmospheric pressure. A part from their role of flow grids these strips joined together can become part of the core structure.

The calculation of K_∞ for a typical boiling water reactor fitted with spiral shows that the extra enrichment required to compensate for the presence of the absorbant material of the strip is very small (below 0.1 p. 100 for Zircalloy strips 0.4 mm thick).

A boiling reactor core designed to make best use of the spiral strips could operate at a volume power greater (by at least 50%) than that of a reactor without these spirals.

An economic study will show the advantageous conditions of use for this type of modification.

Translation provided by courtesy of the submitting Government

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/96
FRANCE
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

IMPROVEMENT INTRODUCED INTO THE THERMAL TRANSFER OF NUCLEAR FUELS BY THE "VAPOTRON" PROCESS

by Dr LE FRANC - Compagnie Française Thomson-Houston - PARIS
Dr BRUCHNER - Allgemeine Elektricitäts-Gesellshaft - FRANCFORT
Dr DOMENJOUD - Société Alsthom - PARIS
Dr MORIN - Communauté Européenne de l'Energie Atomique

The efforts made with a view to reducing the cost of the electric power of nuclear origin are being pursued along different lines, one of them being the increase in specific power (expressed in kW per liter). Progress in this respect consist either in increasing the thermal power produced inside the fuel proper (e.g. by increasing the neutron flux) or in improving the conditions under which this energy will be transferred into the form under which it will be used for transformation into electricity.

In the case of reactors using uranium oxyde, it was feared for a long time that the specific energy released within the fuel might be severely limited by the maximum temperature at the center of the rod. Actually, it was noted that the thermal conductivity of the fuel could be improved and furthermore that a central fusion did not necessarily have any serious effect on the operation. Consequently, the problem of the transmission of this energy to the coolant mixture (water and steam) is once more to the fore.

It is interesting to note that practically all laboratories working on this problem - many of them with considerable means - are exclusively interested in the thermal exchanges occuring on isothermal surfaces (e.g. cylindrical canning or plates). The authors on the contrary have studied the phenomenon of two-phase exchange on non isothermal canning. These particular structures are fitted with ribs or fins on which is set up a stable temperature gradient particularly

Translation provided by courtesy of the submitting Government

favourable to an exchange by boiling. This special type of exchange by boiling was developed up to a point where, under certain precise thermodynamic conditions, remarkable manifestations known as the "Vapotron effect" were obtained.

The authors describe first the results obtained on cans of this type, in various testing facilities designed to reproduce the thermodynamic conditions of a reactor. It will be seen that the performances achieved promise exchange coefficients much higher than those generally accepted. The very great increase in safety will also be illustrated allowing in particular a set-back of the burn-out conditions (often leading to fuel element melting).

They explain how such a system can be made to fit into the present concept of reactors in which the heat transfer is chiefly achieved by boiling, what essential differences it would cause in their structure, taking into account the technological requirements, and finally they indicate a balance sheet explaining the main advantages of the "Nuclear Vapotron system" on technical and economic grounds.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/97
FRANCE
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

THE EFFECTS OF NEUTRON IRRADIATION ON VARIOUS FUELS USED IN GAS-GRAPHITE REACTOR SYSTEMS

by Mmes J. LANIESSE - Ingénieur au SRMA-DIR
J. LEHMANN - Chef de la Section SRMA-SEAU
MM. C. ALLAIN - Ingénieur au SRMA-SEAU
H. AUBERT - Ingénieur au SRMA-DIR
M. ENGLANDER - Chef du Service RMA

(Commissariat à l'Energie Atomique, France)

The behaviour of natural uranium based fuels in form of rods and tubes has been examined after irradiation in several French reactors.

Two categories of uranium alloys have thus been studied : on one hand alloys contaming from 0, 5 to 3 p. 100 by weight of molybdenum, and on the other hand alloys obtained by addition to nuclear purity uranium, of less than 1500 p. p. m. of metallic elements such as iron, aluminium, chromiumand silicon.

The tests have been carried out with two types of elements : one canned with a magnesium alloy and cooled by carbon dioxide under pressure (in G 2, G 3), the other canned with aluminium and cooled by heavy water (in EL 3). Each of these elements had the same outside form as the usual elements in these reactors, but the enrichment and thickness of the fuel were adapted in order to meet different irradiation conditions.

The post-irradiation examinations described, thereafter involved techniques of metrology, of radiography, of density measurements before and after isochronic annealings, as well as macrographies or optical and electronic micrographies.

Translation provided by courtesy of the submitting Government

The experimental results thus obtained reflect the changes in the internal and external structures of the full-size fuel elements ; they are discussed in terms of the combined effects of pressure, temperature and burn-up on the morphological state, on the isotropy and on the mechanical characteristics of the initial polycrystalline aggregate.

An analysis of the results makes it possible to deduce the various advantages of un-enriched metallic fuels designed to operate with a hot-point of around 600°C in reactors of the gas-graphite type, and for a length of time corresponding to an output of 4500 MW day/metri ton.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/98
FRANCE
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

LONG TERM DEVELOPMENTS IN IRRADIATED NATURAL URANIUM PROCESSING COSTS

OPTIMAL SIZE AND SITING OF PLANTS

by MM. L. THIRIET - C.E.A.
C. OGER - Saint-Gobain Nucléaire
P. De VAUMAS - Saint-Gobain Nucléaire

(Commissariat à l'Energie Atomique, France)

The aim of this paper is to help solve the problem of the selection of optimal sizes and sites for spent nuclear fuel processing plants associated with power capacity programmes already installed.

Firstly, the structure of capital and running costs of irradiated natural uranium processing plants is studied, as well as the influence of plant sizes on these costs and structures. Shipping costs from the production site to the plant must also be added to processing costs. An attempt to reach a minimum cost for the production of a country or a group of countries must therefore take into account both the size and the location of the plants. The foreseeable shipping costs and their structure (freight, insurance, container cost and depreciation), for spent natural uranium are indicated.

Secondly, for various annual spent fuel reprocessing programmes, the optimal sizes and locations of the plants are determined. The sensitivity of the results to the basic assumptions relative to processing costs, shipping costs, the starting up year of the plant programme and the length of period considered, is also tested.

- this rather complex problem of a combinative nature, is solved through dynamic programming methods.

Translation provided by courtesy of the submitting Government

- It is shown that these methods can also be applied to the problem of selecting the optimal sizes and locations of processing plants for M T R type fuel elements, related to research reactor programmes, as well as to future plutonium element processing plants related to breeder reactors.

Thirdly, the case where yearly extraction of the plutonium contained in the irradiated natural uranium is not compulsory is examined ; some stockpiling of the fuel is then allowed some years, entailing delayed processing. The load factor of such plants is thus greatly improved with respect to that of plants where the annual plutonium demand is strictly satisfied. By including spent natural uranium stockpiling costs an optimal rythm of introduction and optimal sizes for spent fuel processing plants are shown, different from those in part two.

The indirect effect of these reprocessing programmes on the availability of plutonium, and therefore on the possibility of undertaking plutonium burning reactor programmes, must be taken into account.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/89
FRANCE
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

ESSENTIAL FACTORS IN A BALANCED NUCLEAR ECONOMY (THERMAL AND BREEDER REACTORS)

by R. GIBRAT - Directeur Général du Groupement pour l'Industrie atomique INDATOM (PARIS - FRANCE)

The report by the U. S. Atomic Energy Commission to President KENNEDY on November 20 th, 1962 started a strong world-wide movement to consider and discuss nuclear economy on a long-term basis. There is general agreement to day that this will consist of thermal and breeder reactors, at least for a long time. Meanwhile earlier articles such as that of MM. ANDRIOT and GAUSSENS, at the operational research Conference at Aix-en-Provence (September 1960), did not attract attention when they appeared. They are now being carefully studied.

For our part we have carried out various investigations during 1963 in an attempt to analyse the future as far as possible, demonstrating for example the essential fact that the higher the growth rate of the total nuclear programme, the more difficult it will be for breeder reactors to take a part. It is only progressively however that we realised the essential role of the following three factors in the development of a balanced economy.:

delays in treatment and manufacture loss of fissile materials in the cycle, hold-up in the inventories.

This paper sets out the equations corresponding to these problems, fixes their dependence as a function of the growth rate of the total nuclear economy, and calculates the degree of penetration of breeders in the overall economy for reasonable values of the various parameters.

For one particular case we have been able to show from these calculations that to vary the net plutonium production of a breeder by 10 p. 100 is

Translation provided by courtesy of the submitting Government

equivalent, if the extent of penetration of a breeder in a given nuclear economy is to be kept constant, to :

- increasing by eleven months the time the load remains in the pile,
- or decreasing the neutron equilibration period by two months,
- or decreasing the treatment, manufacture etc... time by one month,
- or decreasing the percentage of losses per cycle by 2 p. 100,
- or finding the same degree of penetration for a 4,2 p. 100 lower growth rate.

An organisation for distributing research funds could thus divide its efforts in various directions by making use of the formulae presented in this report.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/100
FRANCE
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

RECENT PROGRESS IN FISSION PRODUCT SEPARATION

by M. A. RAGGENBASS

(Commissariat à l'Energie Atomique, France)

I - Separation of caesium 137

Successful experiments have been done on the method described at Geneva in 1958.

The process has been considerably improved :

1° - Initially, the caesium phosphotungstate precipitate was leached barium hydroxide in the centrifuge and this was followed by a distillation of ammonia in a concentrator. The barium hydroxide was then eliminated by carbonate precipitation and centrifugation.

It has been proved that the ammonia distillation could be replaced by its evaporation during centrifugation, thus eliminating the need of a concentrator.

It was then possible to carry out the carbonation on the solide-liquid mixture produced by the baryta water leaching.

2° - In applying the above process to the treatment of solutions derived from uranium molybdenum fuels, concentrating is to be recommended in order to hold the molybdenum in solution by complexing it with phosphoric acid.

This complexing process provides a suspension of zirconium phosphate and ammonium phosphotungstate. These are separated by passing into a basic medium which precipitates the zirconium oxide, then turning back to an acid medium ; the end of the treatment remains unchanged.

3° - Studies carried out in several countries on the exchange properties of hetero-polyacid salts have always met with difficulties as a result of the poor

Translation provided by courtesy of the submitting Government

mechanical properties of these substances. This difficulty has been overcome by wrapping the ammonium phosphotungstate in a zirconium phosphate matrix.

The exchanger obtained possesses :

- satisfactory mechanical properties,
- a capacity of 0.1 milliequivalent per gram in concentrated nitric acid solution.

It can be eluted and regenerated by a solution of an ammonium salt.

II - Separation of strontium 90 and rare earths.

The procedure for recovery of these various fission products is briefly the following :

- extraction of rare earths by di-2-ethyl hexyl phosphoric acid into dodecane at pH 2, the chemical impurities being complexed by citric acid.
- extraction of most of the magnesium at pH 4 by the same solvent, the solvent being satured with magnesium in order to avoid as far as possible the extraction of strontium.
- extraction of strontium at pH 4.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/101
FRANCE
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

THE USE OF BERYLLIUM AS A CANNING MATERIAL THE PROBLEMS ARISING FROM THE BRITTLENESS OF THE METAL AND THEIR PRESENT SOLUTION

(Commissariat à l'Energie Atomique France)

by M. WEISZ, J. MALLEN, J.M. DUPOUY and Y. ADDA

Beryllium has been considered as a canning material for reactors of the EL 4 type (CO_2 cooled, operating at 600°C). Two cases may be envisaged : either the can is not deformable under the exterior forces, but in this case a poor ductility may be tolerated, or else, the creep resistance is not sufficiently high ; the ductility should then be high at operating temperatures and also at room temperature for thermal cycling.

It became rapidly obvious that "ordinary" beryllium had not a sufficient creep resistance and it appeared difficult to increase it for EL 4 use (5 kg/mm^2). Other non-metallurgical factors also contributed to discard this approach. The second possibility was this considered and an attempt was made to increase the ductility of the metal which was for two low. The first objection was to determine to what extent this ductility depended on the purity.

It is not yet possible to say whether the brittleness at ordinary temperatures is due to some low concentration of a particular impurity. In the purest beryllium, which can be obtained by distillation vacuum fusion zone melting or a combination of these methods the critical shear stress of the two possible slip modes (0001) and $(10\bar{1}0)$ along $(11\bar{2}0)$ are even more different than in the less pure metal. This means that in the polycristal, the grains which can deform are those which are favourably oriented for basal slip and the fracture mode is still a cleavage along a basal plane. Neither from a theoretical nor a practical point of view has purification led to a solution to the problem of room-temperature brittleness.

Translation provided by courtesy of the submitting Government

The lack of ductility observed around 600°C is undoubtedly due to the presence of impurities and inclusions (oxides). The solution is to use cast metal of industrial purity (or thermally treated in order to modify the impurity distribution) and to eliminate sintering as a production process.

Since, on the other hand, the intrinsic problem of the low-temperature brittleness was not solved, an attempt was made to overcome it by a stricter control of metallurgical factors. It has thus been possible to increase by a factor of 10 the low-temperature ductibility of tubes (of commercial purity beryllium or even of corrosion-resisting alloys).

This is sufficient for the present purpose. Experiments under way will show whether this is still true after exposure to significant irradiation doses.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/102
FRANCE
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

EXPERIMENTAL METHODS OF REACTOR PHYSICS

by MM. D. BRETON
P. LAFORE

(Commissariat à l'Energie Atomique, France)

This paper is a synthesis of various experimental methods in use with the reactors of the Commissariat à l'Energie Atomique.

The main techniques used are mentioned and the difficulties encountered and the accuracy obtained are particularly dwelt upon. The application of these various methods to reactors in order to obtain specific results is also indicated. This paper consists of five parts.

I - General methods (J. CHEROT, B. SAUTIEZ)

Macroscopic and microscopic flux distribution (anisotropy effect), power distribution, etc ...

II - Kinetic measurements (J. LALANDE, M. SAGOT)

a) pulsed neutron technique : apparatus and accuracy ; application to λ_t and to antireactivity measurements ; application to graphite, light water and beryllium oxide.

b) oscillation techniques (R. VIDAL) : equipment and accuracy ; application to the measurements of effective cross sections and resonance integrals.

c) fluctuations : apparatus and technique of measurement.

Translation provided by courtesy of the submitting Government

III - Poison methods (J. BAILLY, Y. GIRARD, P. LOURME)

Description of methods for introducing and extracting the poison, difficulties encountered with light and heavy water, measurement of temperature coefficients and anti-reactivity.

IV - Spectra measurements (C. CLOUET d'ORVAL, P. DELATTRE)

Choice and development of foils, problems of measurement, application to spectral measurements for thermalization studies, application to dosimetry.

V - Experimental shielding measurements (J. BRISBOIS, M. LOTT, J. RASTOIN)

The technique and apparatus recently developed in this field are presented.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/103
FRANCE
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

THE REACTOR CABRI

by MM. AILLOUD
MILLOT

Ingénieurs au Département des Etudes de Piles
Centre d'Etudes Nucléaires de Cadarache

(Commissariat à l'Energie Atomique, France)

It has become necessary to construct in France a reactor which would permit the investigation of the conditions of functioning of future installations, the choice, the testing and the development of safety devices to be adopted. A water reactor of a type corresponding to the latest C. E. A. constructions in the field of laboratory or university reactors was decided upon : it appeared important to be able to evaluate the risks entailed and to study the possibilities of increasing the power, always demanded by the users ; on the other hand, it is particularly interesting to clarify the phenomena of power oscillation and the risks of burn out.

The work programme for CABRI will be associated with the work carried out on the Americain "Sperts" of the same type ; during its construction, very useful contacts were made with the American specialists who designed these reactors.

A brief description of the reactor is given in the communication as well as the work programme for the first years with respect to the objectives up to now envisaged.

Translation provided by courtesy of the submitting Government

Rough description of the reactor.

CABRI is an open core swimming-pool reactor without any lateral protection, housed in a reinforced building with controlled leakage, in the Centre d'Etudes Nucléaires de Cadarache. It lies alone in the middle of an area whose radius is 300 meters long. Control and measurements equipment stand out on the edge of that zone. It consumes "MTR" fuel elements. The control-safety rods are propelled by compressed air. The maximum flow rate of cooling circuit is 1500 m^3/h. Transient measurements are recorded in a RW330 unit.

Aims and work programme.

CABRI is meant for :

- studies on the safety of water reactors :

- for the definition of the safety margins under working conditions : research of maximum power at which a swimming-pool reactor may operate with respect to a cooling accident, of local boiling effect on the nuclear behaviour of the reactor, performances of the control and safety instruments under exceptional circumstances ...

- experimental investigations on power excursions linked with precise initial conditions : the aim of this work is to define the basis for theoretical research, and the limits beyond which the risks of explosion cease to be negligible. The research work will be done so as to enable checking with outside reactor experiments and to continue them in the explosion field.

- studies of the behaviour of the reactor control-instrumentation.

- experimental investigations related with transient operation with initial short life (study of boiling, temperature measurements, vacuum pressure and fraction ...) with the aim of defining the hypotheses of a theory on swimming-pool reactor kinetics related to heat transfer phenomena.

- investigations of the behaviour of fuels in reactors (these experiments are planned to be carried out in loops)

Preliminary experimental results.

CABRI went critical on the 21st December 1963. The first transient experiments are expected for March 1964.